 STAT

Del

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100

Third United Nations
International Conference
on the Peaceful Uses
of Atomic Energy

A/CONF.28/P/313
USSR
May 1964
Original: RUSSIAN

Confidential until official release during Conference

OPERATING EXPERIENCE OF THE ATOMIC POWER PLANT ON THE ICEBREAKER "LENIN"

Afrikantov

J.J. Aphrikantov, N.M. Mordvinov, P.D. Novikov, B.G. Pologikh, A.K. Sledzjuk, N.S. Khlopkin, N.M. Tsarev

INTRODUCTION

The atomic icebreaker "Lenin" (see Fig.1) joined the Soviet Arctic fleet on December 3,1959. Since that time she has participated annual arctic navigation. By the end of 1963 the icebreaker had covered approximately 60,000 miles, about 40,000 of which in ice field. In conjunction with other icebreakers the "Lenin" has piloted along the Northern Sea Route more than 300 ships. In October of 1961 the icebreaker "Lenin" brought to the starting point the personnel and equipment of the drifting station "North Pole 10".

The use of nuclear power plant (NPP) made it possible to build the icebreaker of great power and sailing autonomy, with a hull of limited sizes, that provide sufficient icebreakability and manoeuvrability. To combine these advantages for an icebreaker with a power plant using chemical fuel is impracticable. In addition, NPP application provides operational advantages of no small importance that include possibilities to maintain practically stable displacement of the icebreaker and to operate it at full power during prolonged period of time in combination with more reliable use of the ship under arduous ice conditions.

The use of the powerful icebreaker along the Northern Sea Route contributes to substantial increase of a ship piloting rate under ice conditions and to navigation time extension. In this case forced winter drifting of transport ships in ice fields is excluded, and the number of accidents and the possibility of ships being lost under ice pressing conditions are also brought down.

In this respect the successful piloting of ships through the Arctic region in 1960 when the icebreaker "Lenin" precluded the destruction menace to a number of the ships in the Vilkitsky strait is extremely significant.

The icebreaker "Lenin" operation over a long period of time under such conditions which are classified in marine practice as the heaviest has shown that the nuclear plant is stable and easily controlled over the whole range of sharp load changes and meets all power requirements of the ship. Vibration-and shock-proof equipment of NPP provides reliable operation of the plant under both ice blows and pitching and rolling.

Now the icebreaker reactors are running with the second fuel charge. The reactors were refuelled in spring of 1963. Each of the three reactors operated with the first charge more than 11,000 hours and produced 430,000 – 490,000 thermal megawatt-hours. An average burn-up per a reactor core was 11,000 – 13,000 MWd/tU and the maximum about 30,000 MWd/tU. During this period fuel elements operated in the primary circuit water for approximately 30,000 hours.

The icebreaker's reactors have shown stable running at various power levels, including the maximum of 90 MW. The simultaneous operation of the three reactors with an output of 65 MW each has provided the icebreaker's full horse-power of 44,000 S.H.P.

GENERAL OPERATIONAL CHARACTERISTIC OF NPP

The atomic power plant is about 3100 tons of gross weight including the biological shield and is designed to produce 360 tons of steam per hour at a pressure of 28 kg/cm^2 and a temperature of 300 – 310°C. Figure 2 shows the general view of the plant.

The principal scheme of NPP plant is given on figure 3. The primary circuit consists of three separate sections, each including the following equipment: a reactor, two steam generators, four main circulating pumps, two emergency circulating pumps, four pressurizers and two ion-exchange filters. Each section has two loops – bow and stern ones, that is convenient for both plant operation and piping system and equipment maintenance.

During basic working conditions of a reactor two loops are running. In this case one main circulating pump of each loop is in the "on" condition and another is in the "ready-for-action" condition.

At about 50-megawatt power level the scheme provides reactor operation with one running loop when one main and one emergency circulating pumps of the loop are on. The course of NPP operation has confirmed its designed characteristics that can be seen from the following data for the icebreaker operation at full power.

Note: fraction numerator refers to a bow loop, fraction denominator refers to a stern loop.

The primary-circuit pump rate has proved to be slightly higher than the designed value. In accordance with it water heating in reactor was reduced as it can be well seen from figure 4, where the calculated (1) and experimental (2) graphs of the reactor inlet and outlet temperatures against reactor power are shown.

In 1961 reactor temperature conditions were changed to approach them to self-regulating when a temperature effect is in practice fully compensated by the Doppler effect.

313

Table 1

	Designed values for 65 MW	Operation data for reactors No.1, No.2, No.3		
Water flow in primary circuit loops, t/hr	415	435/430	458/467	435/453
Reactor outlet temperature for loops, °C	317	311/311	312/313	311/313
Reactor inlet temperature for loops, °C	261	260	261	260
Steam output for loops, t/hr	43.3/43.3	42/42	47/42	43/46
Steam pressure, kg/cm^2	29	32/31	31.5/30.5	31/31
Steam temperature, °C	307	310/309	308/308	308/308
Power value determined from primary circuit parameters, %	72.3	69.6	75.4	73.1
Power value determined from secondary circuit parameters	68	67.3	71.1	71.3

As one can see from figure 4, reactor outlet temperature at all power levels is practically constant (3) under self-regulating conditions.

During reactor core life the characteristics, corresponding to self-regulating conditions, slightly change due to temperature and Doppler effect. However it does not result in necessitating control system alteration.

The plant design provides the possibility of steam supply for all users from the general ship line: main turbine units, electrical stations, evaporative units, etc. With the aid of valve system this general steam line can be separated into sections with the groups of steam generators from one or two reactors which provide steam separately to the bow and stern echelons of equipment. The experience gained has shown that the operation of all the steam generators for the general steam line has certain advantages over the above-mentioned echeloned operation. Reactor scram during an echeloned steam supply results in stopping steam provision to the users fed from the steam generators of this reactor; while with the general steam line running it turns out possible to maintain the required pressure inside it due to power increase of the other reactors therefore there is no necessity in full disconnection of the steam users.

Owing to this the number of valve system switches under transient and emergency conditions of the plant operation goes down and hence the probability of false operations during the most serious periods of plant running is reduced.

The operational experience has shown sufficient reliability of the icebreaker's emergency measures for providing a regular electric power supply to NPP from two electric stations of 3000 kW each. In case of voltage loss by one of the stations two emergency diesel generators of 100 kW each are automatically started and a 1000 kW-reserve diesel generator is started by hand.

By the proposal of the operational personnel the designed two-boarded relay-contact circuit of power supply for reactor control and safety system has been substituted by a semiconductor valve circuit to up the plant reliability.

313

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100015-4

It should be noted that there has been no cases of failing in safety trips during the whole operation period of the reactors. Investigations have shown that in some cases, for example in the event of a stop of one of the working circulating pumps, there is no necessity in dropping power to zero level, but its fast automatic reduction up to the level of 30% – a so-called second-stage safety trip – – is quite sufficient. In this connection a part of signals which previously caused power drops to a zero level has been transferred to the signal class of the second-stage safety trip. This has increased ship vitality and lowered the number of sharp thermal oscillations of the plant equipment. The automatic power drop up to a zero level has been referred to as "a first-stage safety trip".

All the equipment of the primary circuits of the icebreaker's atomic power plant has operated, including 1963, about 15,000 hours at working conditions (at a pressure of 180 kg/cm^2 and temperatures of 250 – 310°C). Main circulating pumps have run up to 8000 – 9000 hours without any inspection. However some of the pumps failed due to lowering the insulation resistance of stator winding. Emergency circulating pumps have shown reliable operation. The steam pressurizing system has provided high stability in maintaining the primary circuit pressure at stationary power levels and prevented pressure oscillations of more than ±5 atm during transients.

The active corrosion products settling in the lower part of the pressurizers complicate dismantling work and electric heaters' replacement. This kind of work has not yet been carried out. Ion-exchange filters have been able to maintain the required water quality in the primary and secondary circuits of the power plant: specific resistance of 1-2 mo/cm; chlorine ion content of no more than 0.02 mg per litre, and pH value of 6 – 8. Recently the ion-exchange resins КУ-2 and AB-17 have been successfully used in filters.

The addition of hydrodine into the secondary circuit water has been often done. At the beginning of the icebreaker mechanical plant operation several cases of a short-term rise of salt concentration in the secondary circuit feed water took place. But later owing to modification of certain parts of equipment and control system improvement, these phenomena have occurred rather seldom and accompanied by the immediate localization. It is necessary to emphasize the efficiency of double grid plates in the condensers and refrigerators of overboard water, practically excluding sea water pumping.

The NPP steam generators have been reliable and stable under both stationary and transient conditions. But there have been some cases of their pipe systems' unsealing in the course of operation. Measures for detecting and disconnecting a damaged steam generator were taken rather orderly and fast. Owing to this only a short-term rise in water activity of the secondary circuit with the background levels being exceeded by several times has been observed.

Some cases of drop leakage through the seals of main slide-valves due to seal packing drying have taken place in the primary circuits' valve system. The packing has been replaced by a higher-quality one but it has been necessary to tighten seals in every 1500 – 2000-hour intervals. The sylphon valves of the primary circuit drain system have proved to be insufficiently reliable. The annual inspection of these valves has been required.

313

The actuating mechanism of the reactor control and safety system has shown reliable and stable operation and is beyond essential reproves.

The biological shield surrounding reactors, equipment and primary circuit piping lines has proved to be sufficient, it has revealed no failures due to shock loads when the ship sailing in ice fields and during storms. The highly active slurry has leaked into some pulse tubes coming out from the lower piping points and caused rise in gamma-radiation levels at the places where these tubes pass beyond the biological shield and go to the data units of monitoring devices and meters. This has necessitated to arrange additional local shielding.

The NPP maintenance work has revealed the necessity to expand the sanitary treatment block at the entrance to the strict regime zone and provisional stores for liquid and solid wastes. Therefore the sanitary block and stores have been properly re-equipped. The radiation monitors available provided a proper control of the radiation situation on board the icebreaker and the activity level in the atomic plant process circuits. In the course of plant operation a part of these dosimeters has been replaced by new improved ones. The icebreaker's active-gas control system has been modified.

The individual-control results for the personnel operating the atomic power plant have shown that an integral irradiation dose for the overwhelming majority of controlled persons does not exceed one-third or a half the maximum permissible dose of 5 rem rads.

Only some persons who performed radiation-hazardous work connected with premises' decontamination where the primary circuit water leaks occurred have received doses close to the maximum permissible ones (3, 4).

REACTOR CORE NEUTRON-PHYSICAL PARAMETERS

The reactor design description is given in a report /1/. The core consists of 219 fuel assemblies passing through the knots of a regular lattice with a 64-mm pitch (See figure 6); it is of 1.6-m height and 1-m equivalent diameter.

The first fuel charge of each reactor contained 80 kg of uranium-235. The reactor core is designed for 200-day run at a maximum power of 90 MW. One-third of the initial fuel charge deteriorates during this period. Multiplication factor original reserve complying with this rather high burn-up ($\rho = 14\%$) has been decreased by two-fold due to boron-10 addition into the channel casing tubes in quantity of 92 grams per a reactor for the first fuel charge. Boron is spread over a reactor core nonuniformly: with decreased concentration from a reactor axis to periphery; the outer row assemblies contain no boron.

A fuel assembly (see figure 6) contains a 36-cylindrical fuel-element bundle fitted with a spacing frame. Fuel elements are 6.1 mm in outer diameter and minimum element spacing in assembled bundle is 1.5 mm. Element cans are 0.75 mm thick. Fuel assembly casing tubes, fuel elements cans and spacing frames for the first charge were made of a zirconium alloy. Sintered uranium dioxide pellets of a 5% average enrichment are used as a fuel.

In-loop test results could not produce experimental information on the process dealt with fuel burn-up at the real positions of the compensation mechanism and neutron field distortions. These data could be derived from reactor core fuel irradiation as a whole.

3 ; 3

The reactor core development was based upon calculated data. All the calculations were carried out with a quick-acting electron computer using two standard one-dimensional programs (radial and high) in this case an effective multiplication factor and a radial neutron leakage were assumed to be constant.

A calculation basis for the multiplication parameters in process channel lattice was an approximation technique founded on introducing empiric corrections derived from analysis of a great deal of critical experiments into a classical thermal-reactor model.

Reactor operation on the icebreaker was preceeded by reactor-core investigation at a testing jig. During these tests both the reactor core design and the design of reactor control and reactivity compensation systems were ultimately mastered.

Figure 7 shows the level of the compensation system insertion into a reactor core as a function of energy generation in reactor No.1 during its long-term operation at a power range from 40 MW to 60 MW.

The design curve (1) is determined from one-dimensional axial program. It is in good consistency with experimental points corresponding to the compensation system positions during reactor long-term operation at stationary power levels. Thus in spite of the design model proximity a theoretical description gives rather satisfactory results.

Fast insertion of the compensation system at the beginning of the reactor lifetime is attributed mainly to boron burn-up effect: reactivity excess caused by boron burn-up is not compensated by reactivity loss due to uranium-235 burn-up. The equilibrium occurs only in 79 days after reactor maximum-output operation.

The right branch of the curve is less crooked due to liberation of the fuel assembly slightly-burned regions during the compensation system removal out of the reactor core. On the figure 7 one can see some points corresponding to the reactor compensation system positions under various poisoning at temperatures from 40°C to 80°C and maximum reactivity values (curve 2).

During the reactor lifetime its minimum subcriticality with the compensation system being positioned on the lower end switches (3) and with the control and safety rods being fully withdrawn is equal to 1 – 1.5%. This provides reliable reactor blocking.

Radial energy distribution before the reactor operation under working condition was determined from the activity level of the process channels exposed to radiation within the core at a temperature about 20°C and a minimum power level. The radial peaking factor for this case proved to be 1.2 (see figure 8). The design curve (see figure 9 2a2) is referred to hot-poisoned-reactor peaking in process channels with ignoring the effect of thick-walled steel casings for control and safety rods which are located around the reactor axis. This accounts for the divergence of calculated and experimental data for the central process channels.

During reactor operation all the changes in the energy-generation field shape were controlled by resistance-thermometer measurements of temperature drops in fuel assemblies. By the time of reaching maximum reactivity (at an energy generation about 160,000 MW/hr) radial energy-release peaking had become maximum – about 1.42 (figure 8 "b") due to boron burn-up in reactor central zone.

313

In an energy-generation range from 160,000 to 420,000 MW/hr radial energy-generation peaking is slightly improved due to more intensive burn-up of uranium-235 in the core centre and achieves a value of 1.28 (curve C).

Divergence between the calculation and experiment arises in this case from errors in temperature measurements and from approximate regard for an axial component when making calculations.

The axial peaking factor values for thermal neutron field and energy generation at various insertion levels of the reactivity compensation system and under various operation conditions of reactor No.1 are given on figure 9. The design curve "a" for a neutron field peaking factor corresponding to a cold not-poisoned reactor at the beginning of the reactivity lifetime period is in good consistency with experimental points. All the measurements were carried out with portable scaling counters being displaced within the steel casings for automatic controls. A change in the axial peaking of thermal neutron field during reactor lifetime and its operation with a 50-MW capacity are shown by the curve "b". The experimental points represented by this curve were determined from copper wire activation at a power range from 18 MW to 55 MW. The curve "c" shows the dependence of axial energy-generation peaking.

It should be noted that nuclear fuel burn-up along a reactor axis goes more uniformly than it is seen from the energy-generation peaking factors shown on figure 9.

Due to maximum neutron flux field travelling along a reactor axis fuel burn-up curve has a plateau (about 8 cm long), in the central region with a burn-up level as high as 47% of the initial amount of uranium-235.

The curves illustrating the changes in reactivity temperature effect are given on figure 10. These curves apply to the beginning of reactor core lifetime (a) and the time of achieving an energy-generation level about 430,000 MW/hr (b). The reactor core geometric factor effect is excluded; in both cases the compensation system insertion levels were practically identical.

The reactor core warming up was provided from both an outside heat source by steam supplying to the steam generators from the secondary circuit side and by the primary circuit coolant circulating and also from the own heat when operating with an output of 1 – 10 MW. Figure 11 shows reactivity changes for the reactor lifetime start and energy generation about 430,000 MW/hr as a function of a power level.

Up to a 10%-power level the data were received due to prompt jumps of positive reactivity of various values and measuring power levels achieved by the reactor.

The Doppler effect values for power levels above 10% were determined at stable coolant circulation, fast up-and-down power jumps and also at a constant moderator temperature and variable water circulation through the reactor. The results obtained were properly corrected to exclude the moderator temperature-rise effect. At the end of reactivity lifetime the Doppler effect increase has been observed that probably arises from changes in the structure and heat conductivity of fuel element core material.

During the icebreaker reactor operation the efficiency of regulators was periodically measured. At the start period of reactivity lifetime the weight for one group of automatic control rods (3 rods) was 0.36%. At the maximum reactivity when achieving energy generation of

313

160,000 MW/hr the control rod group weight rose up to 0.7% and was practically kept at this level to the end of the core lifetime.

This control rod weight increase arises basically from extending neutron fields in the region of their disposal and also from growing the thermal-neutron diffusion length in the reactor core as nuclear fuel and boron are burned up.

REACTOR REFUELLING AND BASE MAINTENANCE

The first overcharge of the icebreaker reactors was performed in spring of 1963. The auxiliary ship "Lepse" was used to carry out this operation. The motor ship "Lepse" is equipped with a spent-fuel storage and refuelling facility including a turning crane of 12-ton hoisting capacity, shield in containers for unloading fuel assemblies and control and safety rods, a guide device for placing an overloading container against a proper reactor cell, etc.

The icebreaker crane was also used for reactor overcharge.

The sequence of reactor refuelling operations is as follows. An overloading container [see figure 12 (1)] is put on the shielding plate of the guide device (2) above an assembly to be overloaded. After that an expansion tong (5) inside the container is lowered by a hand winch to clamp the assembly head. The expansion tong operation is controlled with a periscope (6). The assembly is hoisted by the hand winch into the container, the bottom slide of which is then closed and the container with the assembly inside is carried by the icebreaker crane on board the ship "Lepse" for its loading into the spent-fuel storage.

In case of an assembly being mechanically wedged in reactor cell it is made a move by a jack. The number of such wedges during reactor discharge was rather smallu. An average time of one assembly discharge was about 15—20 minutes. It took about three days to bring about the total reactor discharge.

Investigation of the fuel elements unloaded from the icebreaker reactors has shown that they are in good condition (see figure 13). The fuel elements has revealed no swelling, bending, abrasion signs or notable rod diameter changes. Can surfaces are covered with a thin layer of dark coloured depositions of some microns thick.

Figure 14 illustrates a fuel element cross ground end after a power generation level about 15,000 MWd/tU.

In spite of the fuel element core cracking the space between the core and its can has been however kept. The temperatures in fuel core centre have not been close to the melting temperature of uranium dioxide that is confirmed by the absence of a central hole in a fuel pillet and an anomalous grain growth.

After removing all the fuel assemblies reactors and their primary circuits were washed off.

During reactor charging operation fuel assemblies were pulled down with all absorbers being inserted into the reactor core and neutron density change control being duplicated. It took 6—10 hours to load fuel assemblies into a reactor core.

After charging new modified fuel assemblies the reactors were run critical with the consequent checking of the control rod weights and reactor shield. These new assemblies

313

were of the same geometrical sizes but manufactured by the improved technique. One of the reactors was loaded with steel-encased fuel elements.

The icebreaker reactor refuelling operations were carried out in the region of Murmansk where a moorage is arranged and the minimum equipment required for the icebreaker "Lenin's" maintenance is available.

TRAINING OF OPERATING FORCE

The engineering staff operating the icebreaker atomic plant went through special training at the Makarov Higher Engineer Nautical Institute in Leningrad. After that it worked on probation at atomic power stations. The training and practising courses on reactor control ended in going in for examinations before the State Commission. Later on the examinations to get a working place have been taken annually.

The common operating personnel of the icebreaker atomic plant gained the necessary operational experience at the enterprises of atomic industry.

The first two years of the atomic power plant operation were the years of practising with the plant. This period enabled to reveal all the advantages and disadvantages of the plant, to outline ways for improving the plant control system and to master the transients. At that time the operating personnel of the atomic power plant showed a great piece of creative initiative, put forward a variety of interesting proposals on redesigning the control circuits and some components of the plant. A part of these proposals has been already realized.

CONCLUSION

The long-term operation of atomic heat-generating plant on board the icebreaker has permitted to make a comprehensive valuation of its working capacity under various swimming conditions.

The principal scheme and arrangement of the atomic plant have proved to be successful and the provided reserve equipment to be quite sufficient.

No overirradiation of the personnel has occurred during the whole operating period of the icebreaker. The atomic plant has proved to be so reliable that in order to inspect the equipment only one visit per day was required.

The first experience of installing an atomic power plant on board the icebreaker has proved to be quite successful. The technical expediency for the building of powerful atomic icebreakers to service the Northern Sea Route has been confirmed.

Experimental neutron-physical characteristics of the icebreaker's reactor cores have been obtained by N.A.Lasukov and A.K.Sledziuk and their collaborators. Data derived from control room registers and log-books have been also used in preparing this report.

313

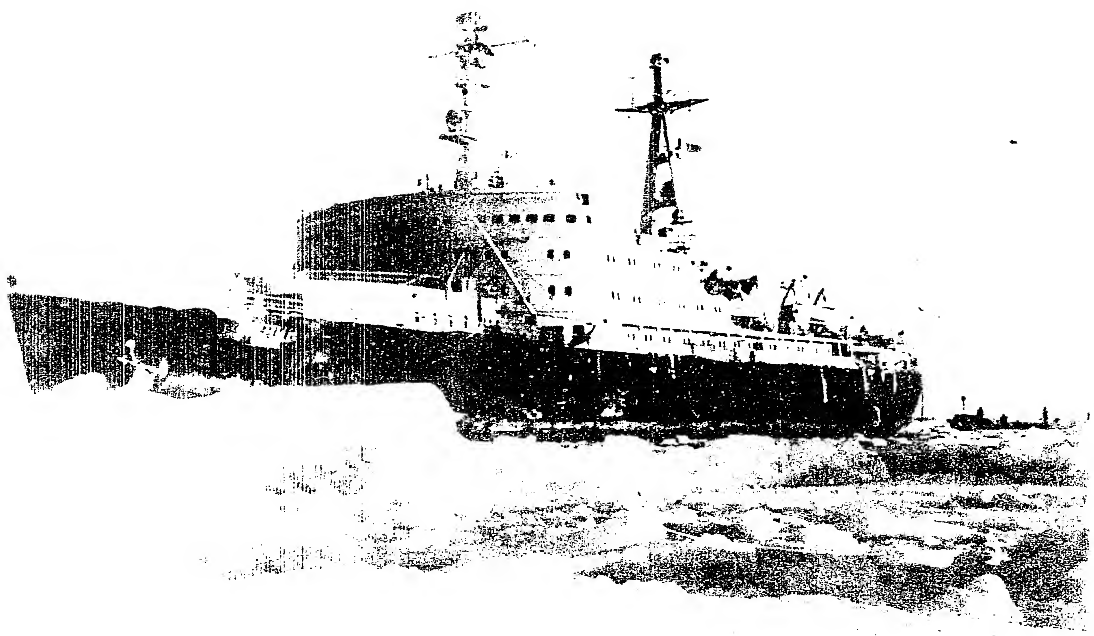

FIG.1. ATOMIC ICEBREAKER "LENIN"

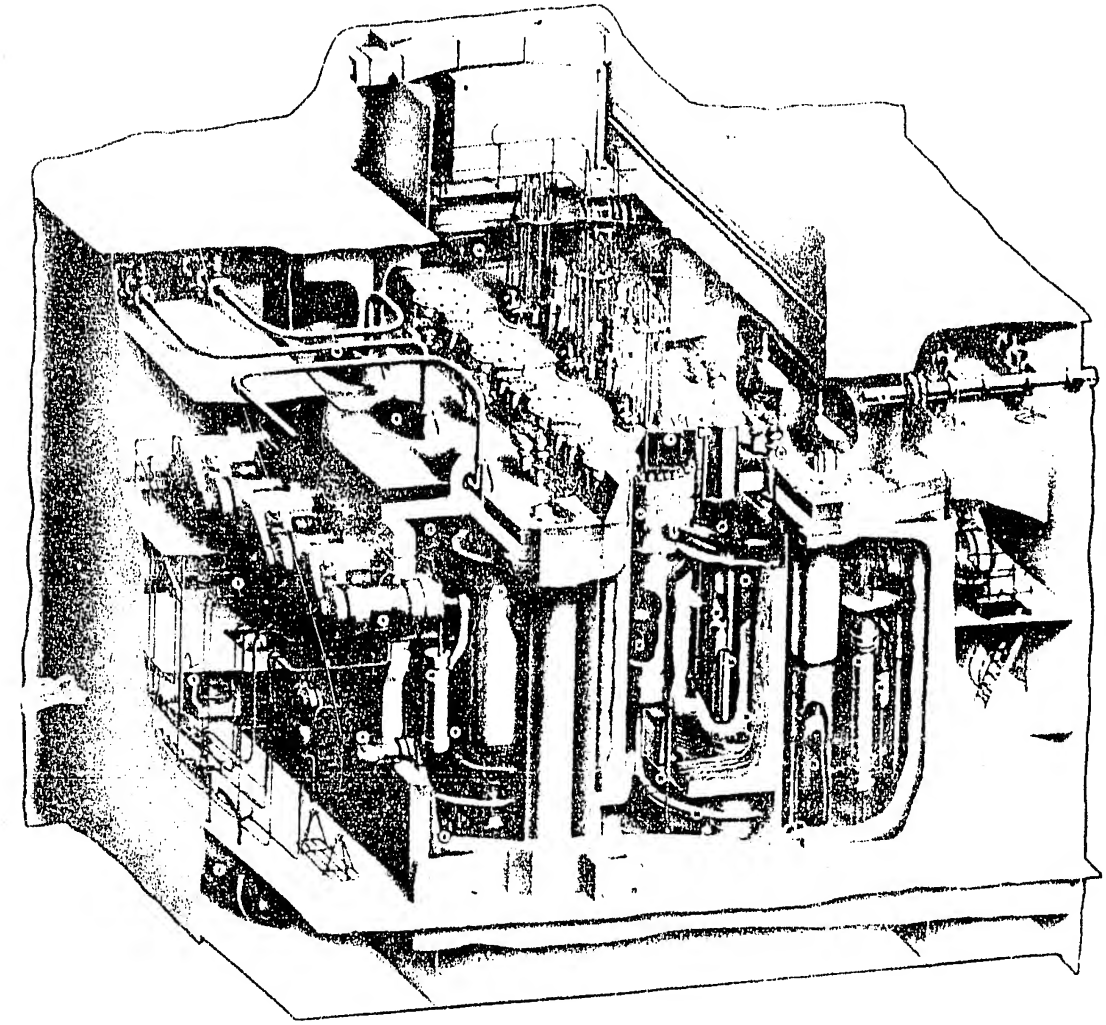

FIG.2. GENERAL VIEW OF STEAM GENERATING POWER PLANT

1 – heat exchanger of III-IV circuit loop; 2 – pump of the internal cooling circuit; 3 – pipeline of outside cooling circuit; 4 – air pump valves; 5 – main circulating pump of the primary circuit; 6 – emergency circulating pump of the primary circuit; 7 – room for the thermal-control data units; 8 – valves of primary circuit drain system; 9 – heat insulation; 10 – primary-circuit filter cooler; 11 – steam generator; 12 – steam generator room; 13 – shield for the steam pipeline outlet; 14 – steam piping; 15 – emergency coolant tank; 16 – control and safety system room; 17 – actuating mechanisms of the control and safety system; 18 – ionization chamber; 19 – carborite; 20 – reactor; 21 – reactor core; 22 – level indicators; 23 – biological shield for the ventilating shaft; 24 – primary circuit piping; 25 – iron-water shield; 26 – equipment storage pit; 27 – primary circuit filter; 28 – iron-water shield tank; 29 – concrete; 30 – primary circuit main pipeline shutter; 31 – piping collector; 32 – hatchway to the steam generator room; 33 – heat exchanger of the internal coolant circuit; 34 – mechanical purification filter; 35 – pressurizer; 36 – electric heaters.

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100015-4

313

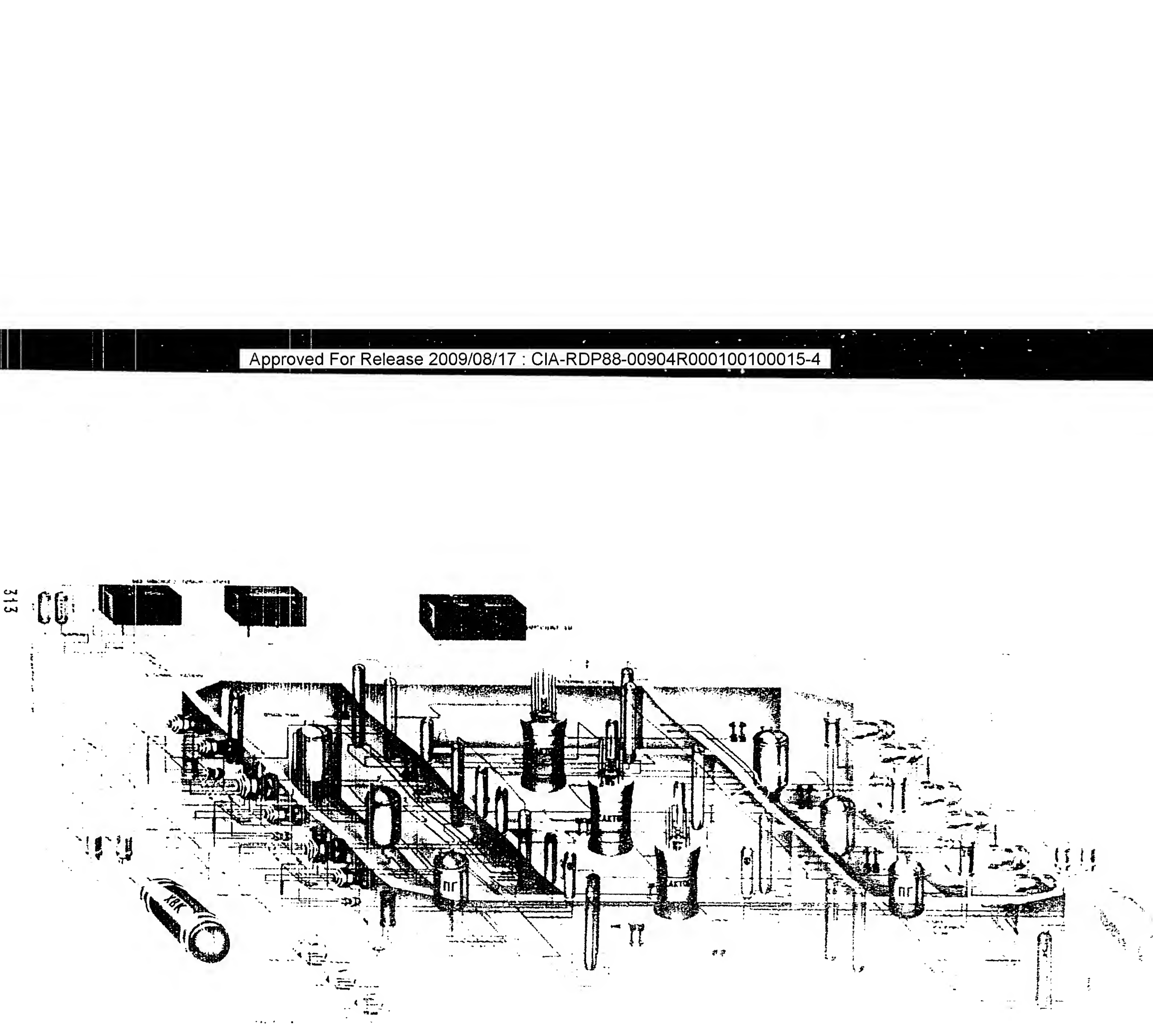

FIG.3. PLANT PRINCIPAL HEAT FLOW DIAGRAM

═══ coolant circuit; SG – steam generator; P – pressurizer; MCP – main circulating pump; ECP – emergency circulating pump; IEF – ion exchage filter; FC – filter cooler; ICCC – internal coolant circuit cooler; ECC – external circuit cooler; FP – feed pump.

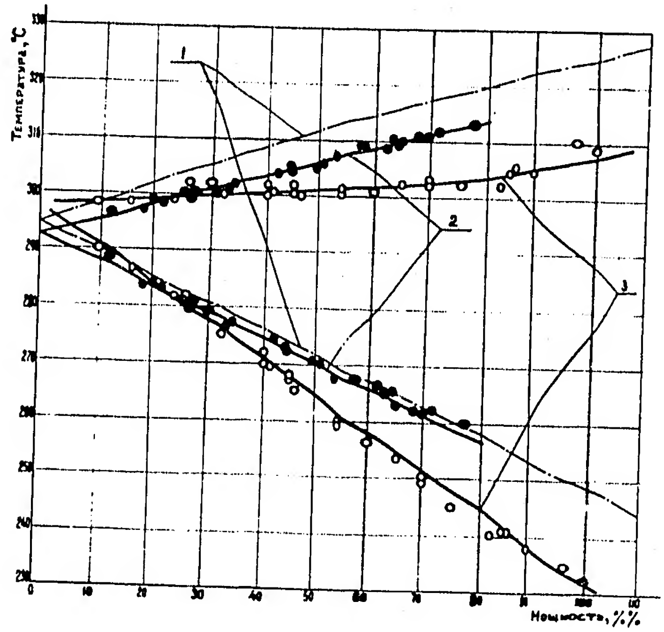

FIG.4. INLET AND OUTLET TEMPERATURES AS A FUNCTION OF REACTOR OUTPUT

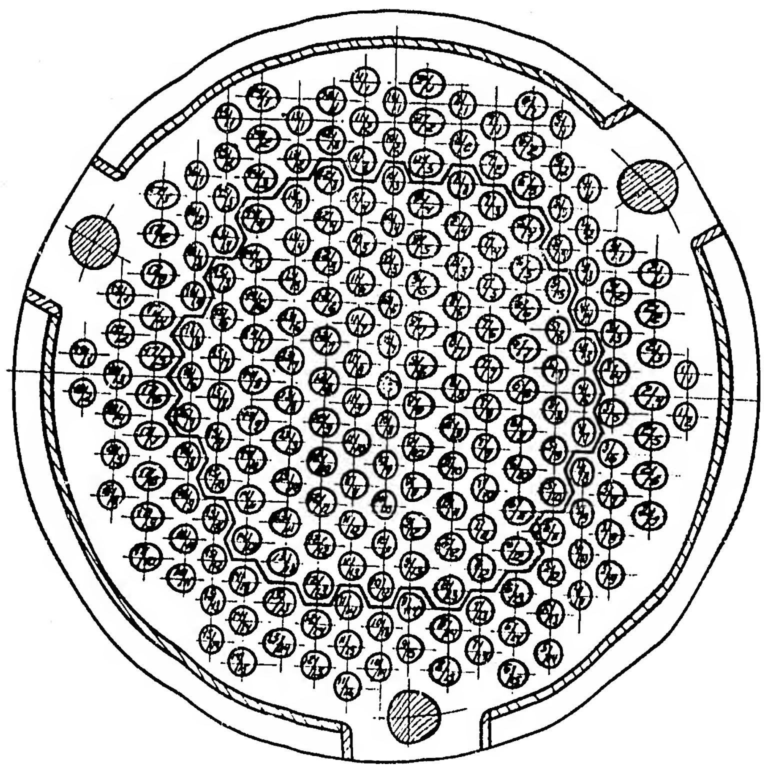

FIG.5. REACTOR CORE CROSS SECTION

313

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100015-4

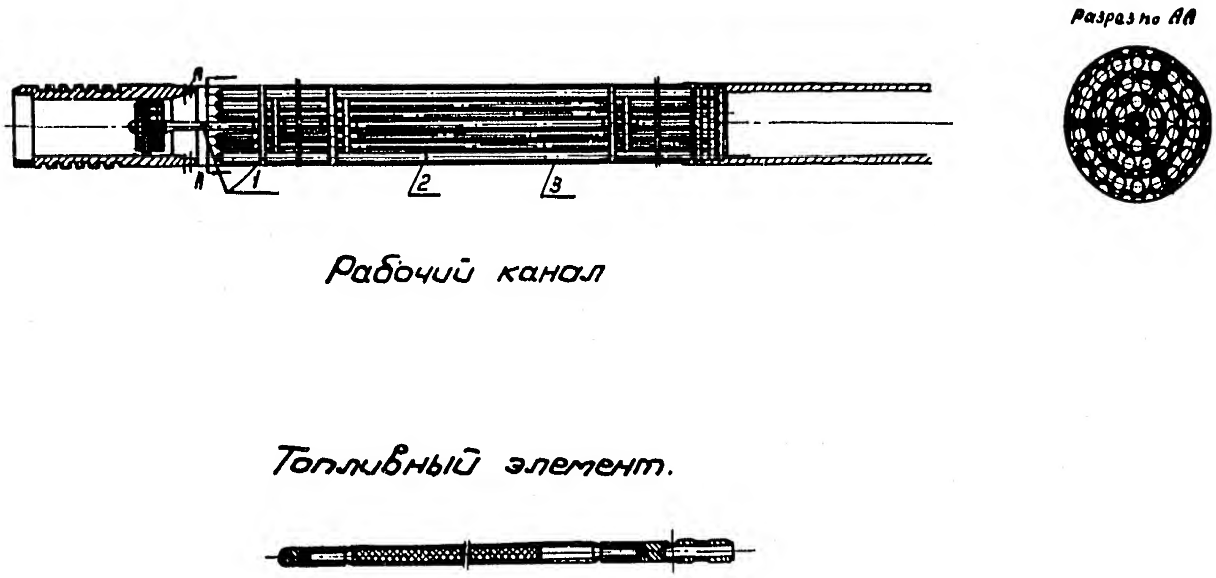

FIG.6. PROCESS CHANNEL AND FUEL ELEMENT LONGITUDINAL SECTION

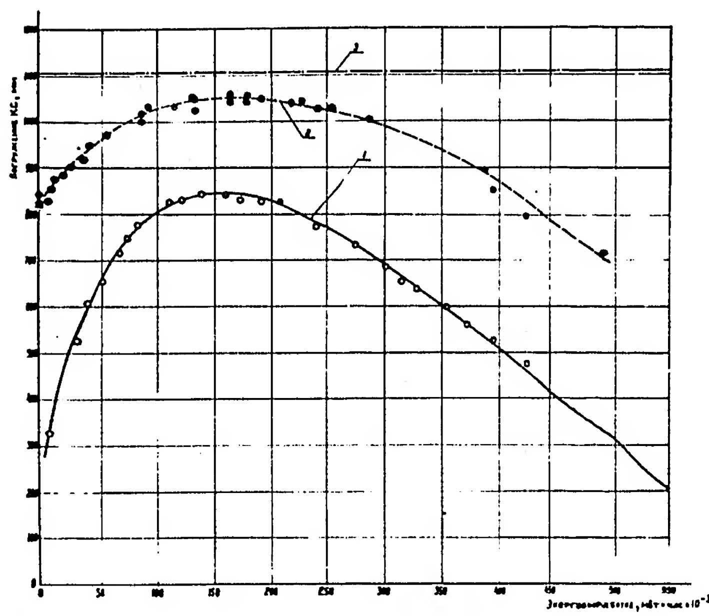

FIG.7. COMPENSATION SYSTEM INSERTION INTO REACTOR CORE VERSUS REACTOR 1 POWER GENERATION

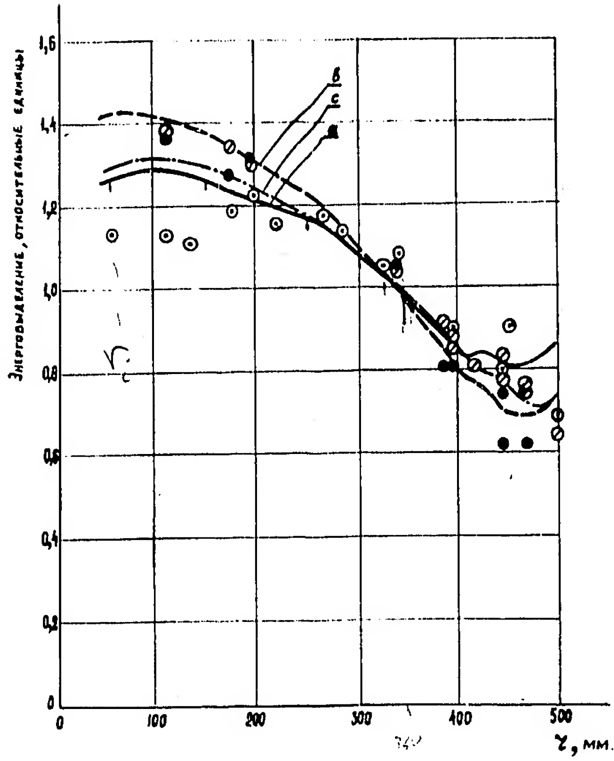

FIG.8. RADIAL ENERGY-GENERATION DISTRIBUTION:

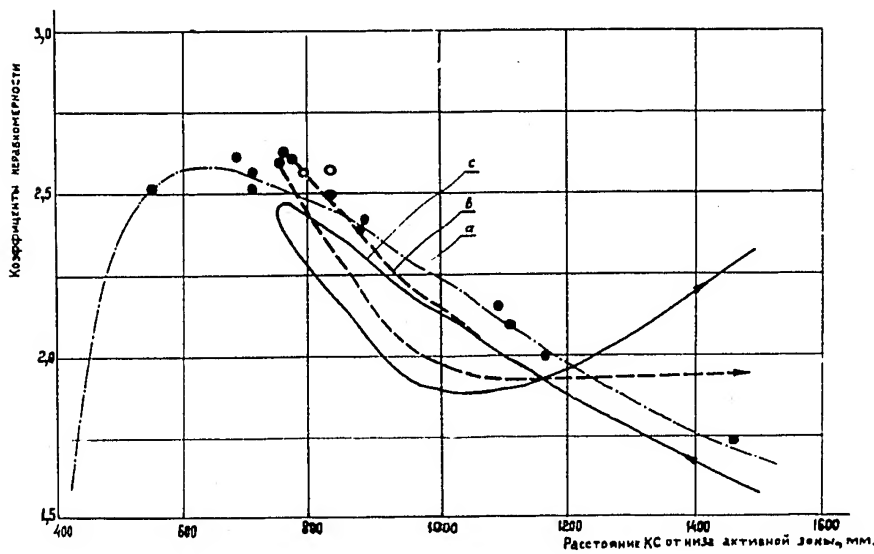

FIG.9. AXIAL PEAKING FACTORS FOR THERMAL NEUTRON FIELD AND ENERGY GENERATION AT VARIOUS POSITIONS OF COMPENSATION SYSTEM

Experimental points:

o – at a minimum controlled power at reactor lifetime start
o – 18 MW power and 60,000 MW/hr energy generation
ø – 55 MW power and 70,000 MW/hr energy generation
● – 54 MW power and 110,000 MW/hr energy generation
ə – 55 MW power and 160,000 MW/hr energy generation

313

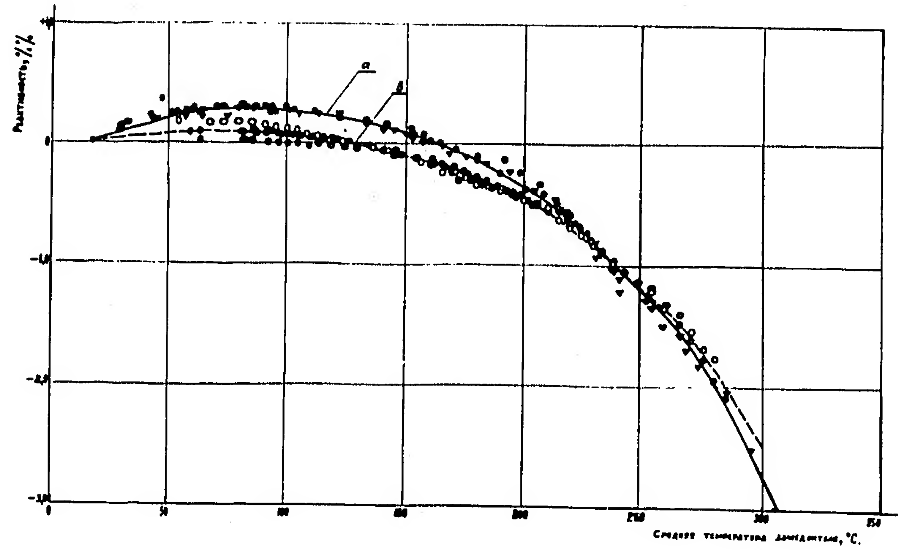

FIG.10. TEMPERATURE REACTIVITY EFFECT

Experimental points during reactor warming up: o, ө – outside source heating;
Δ, ϕ – selfheating;
o – points for reactor removal of after-heat

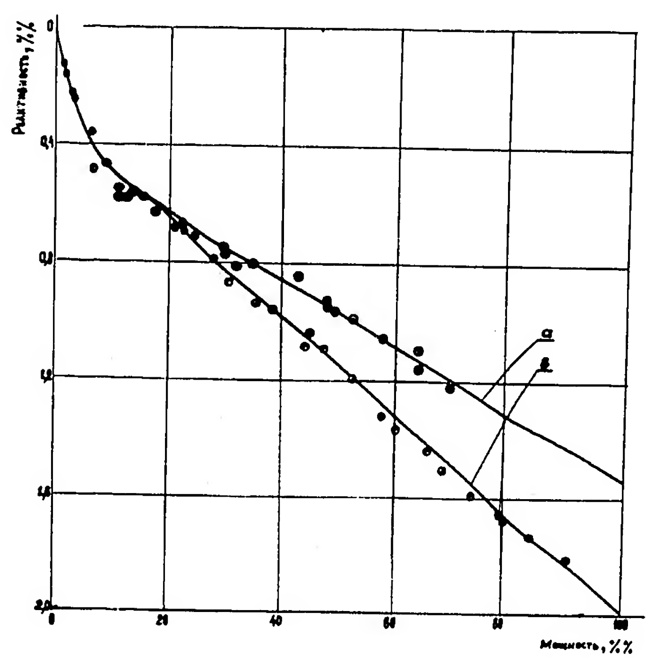

FIG.11. REACTIVITY CHANGES DUE TO THE DOPPLER EFFECT

313

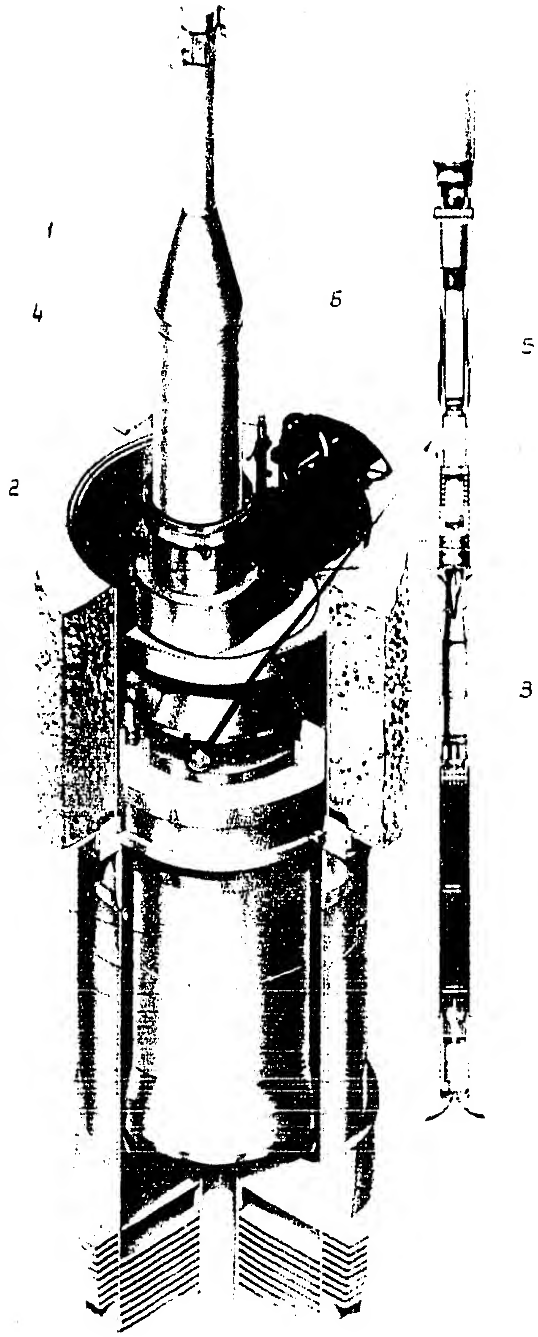

FIG.12. REACTOR WITH THE REFUELLING EQUIPMENT

313

FIG.13. FUEL ELEMENT CHANNEL
WITH THE OUTSIDE CASE REMOVED

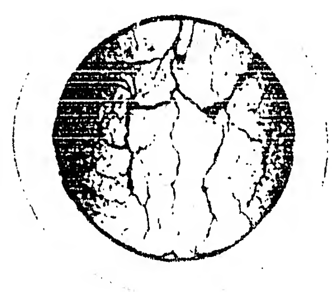

FIG.14. FUEL ELEMENT CROSS
GROUND END AFTER FUEL
IRRADIATION ABOUT
15,000 MWd/tU (25 FOLD
MULTIPLICATION)

313